ACCESSION #: 9204080122
LICENSEE EVENT REPORT (LER)

FACILITY NAME: R. E. Ginna Nuclear Power Plant PAGE: 1 OF 9

DOCKET NUMBER: 05000244

TITLE: Feedwater Transient, Due To Loss of "A" Main Feedwater Pump, Causes Lo Lo Steam Generator Level Reactor Trip
EVENT DATE: 02/29/92 LER #: 92-003-00 REPORT DATE: 03/30/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 097

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Wesley H. Backus Technical TELEPHONE: (315) 524-4446
Assistant to the Operations Manager
COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On February 29, 1992 at approximately 1346 EST, with the reactor stable at approximately 97% reactor power, just subsequent to a main feedwater pump trip, a reactor trip occurred due to Lo Lo level (
Steam Generator (S/G).

The Control Room operators immediately performed the appropriate actions of Emergency Operating Procedures E-0 (Reactor Trip Or Safety injection) and ES-0.1 (Reactor Trip Response). Both Main Steam Isolation Valves (MSIVs) were subsequently closed to limit a Reactor Coolant System (RCS) cooldown and the plant was stabilized at hot shutdown.

The underlying cause of the event was plugged instrument tubing for the "A" Feedwater Pump Seal Injection Differential Pressure (D/P) switch which tripped the main feedwater pump. (This event is NUREG-1022 (X) cause code.)

Corrective action taken was the unplugging of the instrument tubing by flushing and the replacement of one section of tubing and connections. Corrective actions to prevent recurrence are discussed in section V of the text.

END OF ABSTRACT

I. PRE-EVENT PLANT CONDITIONS

The plant was a approximately 97% steady state reactor power with no major activities in progress.

II. DESCRIPTION OF EVENT

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

o February 29, 1992, 1346 EST: Event Date and Time

o February 29, 1992, 1346 EST: Discovery Date and Time

o February 29, 1992, 1346 EST: Control Room operators verify both reactor trip breakers open, and all control and shutdown rods inserted.

o February 29, 1992, 1350 EST: Control Room operators close both Main Steam Isolation Valves (MSIVs) to limit a Reactor Coolant System (RCS) cooldown.

o February 29, 1992, 1358 EST: Plant stabilized at hot shutdown condition.

B. EVENT:

On February 29, 1992 at approximately 1346 EST, with the reactor stable at approximately 97% reactor power, the Control Room received Annunciator Alarm H-11 (Feed Pump Seal Water Lo Diff Press 15 Psi) followed in approximately five (5) seconds by a trip of the "A" Main Feedwater Pump. The Control Room operators immediately entered Abnormal Procedure, AP-FW.1 (Partial Or Complete Loss Of Main Feedwater) and performed the immediate actions (i.e. starting all three (3) Auxiliary Feedwater (AFW) pumps, and decreasing turbine power rapidly to less than 50%.)

TEXT PAGE 3 OF 9

During the performance of these immediate actions, a reactor
trip occurred due to Lo Lo level (
Generator (S/G).

The Control Room operators performed the immediate actions of Emergency Operating Procedure E-0 (Reactor Trip Or Safety Injection) and transitioned to Emergency Operating Procedure ES-0.1 (Reactor Trip Response) when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required.

Both MSIVs were subsequently closed at 1350 EST to limit the RCS cooldown. The closing of the MSIVs mitigated the RCS cooldown and the plant was subsequently stabilized in hot shutdown at approximately 1358 EST.

The Intermediate Range Nuclear Instrumentation Channel N-35, after tracking consistent with channel N-36 down to approximately 1E-10 amps, had its indication continue to drop below 1E-11 amps. The N-35 channel returned to normal (i.e. 1E-11 amps) approximately ten (10) hours following the trip.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED
TO THE EVENT:

None.

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTIONS:

None

E. METHOD OF DISCOVERY:

The event was immediately apparent due to alarms and indications in the Control Room.

TEXT PAGE 4 OF 9

F. OPERATOR ACTION:

After the reactor trip, the Control Room operators performed the actions of Emergency Operating Procedures E-0 (Reactor Trip

Or Safety Injection) and ES-0.1 (Reactor Trip Response). The MSIVs were manually actuated closed approximately four (4) minutes after the trip to prevent further plant cooldown. The plant was subsequently stabilized at hot shutdown. Subsequently, the Control Room operators notified higher supervision and the Nuclear Regulatory commission per 10CFR50.72, Non-Emergency, 4 Hour Notification.

G. SAFETY SYSTEM RESPONSES:

None.

III. CAUSE OF EVENT

A. IMMEDIATE CAUSE:

The reactor trip was due to "B" S/G Lo Lo level (

B. INTERMEDIATE CAUSE:

The "B" S/G Lo Lo Level (
feedwater flow to steam flow (i.e. feedwater flow was approximately one half of steam flow) caused by the tripping of the "A" Main Feedwater Pump.

The tripping of the "A" Main Feedwater Pump was due to the inadvertent operation of the Feedwater Pump Seal Water Differential Pressure (D/P) switch. This D/P switch senses the feedwater pump suction pressure on the low side and the feedwater pump seal injection pressure on the high side. The requirements of differential pressure to protect the pump seals is that the high side be >/=15 pounds per square inch pressure greater than the low side. If the above condition is not met, the main feedwater pump will trip in 5 seconds.

The inadvertent operation of the Seal Water D/P switch was due to the plugging of the high pressure side tubing to the D/P switch, followed by gradual decrease of pressure in the isolated high pressure side tubing to the D/P switch.

C. ROOT CAUSE:

The plugging of the high pressure side tubing to the D/P switch was due to an accumulation of corrosion products that built up over several years in an area of no flow through this tubing (i.e. a dead leg).

The pressure decrease in the high pressure side tubing was most probably due to a combination of plugging of the high side tubing and a slight amount of seepage of fluid from a tubing connection on the high pressure side.

IV. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2) (iv), which requires a report of, "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System (RPS) ". The "B" S/G Lo Lo level reactor trip was an automatic actuation of the RPS.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no safety consequences or implications attributed to the reactor trip because:

o The two reactor trip breakers opened as required.

o All control and shutdown rods inserted as designed.

o The plant was stabilized at hot shutdown.

The Ginna updated Final Safety Analysis Report (UFSAR) Chapter 15.2.6, "Loss Of Normal Feedwater", was reviewed and compared to the plant response for this event. The UFSAR transient is a complete loss of Main Feedwater (MFW) at full power, with only one AFW pump available one (1) minute after the loss of MFW, and secondary steam relief (i.e. decay heat removal) through the safety valves only.
The protection against a loss of MFW includes the reactor trip on Lo Lo S/G water level and the start of the AFW pumps. These protection features operated as designed.

The UFSAR transient resulted in a reactor trip on Lo Lo S/G water level with S/G levels continuing to decrease and pressurizer (PZR) level and RCS average temperature (T sub AVG) increasing until the flow from one (1) AFW pump could remove decay heat at approximately

30 minutes into the event. All parameters then trended towards normal.

The plant transient for this event resulted in a PZR level increase to 56.4% and a T sub AVG increase to 577 degrees F. S/G levels remained in the narrow range throughout the transient. This was due to operator's actions to reduce power, steam dump action and the additional AFW flow.

Based on the above evaluation, the plant transient of February 29, 1992 is bounded by the UFSAR Safety Analysis assumptions.

Following the reactor trip, pressurizer level decreased to 0% but began to increase above 0% within approximately five (5) minutes. This is an expected observed transient. The moderate RCS cooldown did not result in any core voiding. This was confirmed by the Reactor Vessel Level Indicating System (RVLIS), which always indicated a level of 100%.

A slow cooldown occurred during the post trip recovery period. This cooldown was bounded by the plant accident analysis and did not exceed the technical specification limit of 100 degrees F per hour. Additional protection was provided by closure of the MSIVs.

Based on the above and a review of post trip data and past plant transients, it can be concluded that the plant operated as designed and that there was no unreviewed safety questions and that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL
STATUS:

o The S/G levels were returned to their normal operating levels by addition of AFW, subsequent to the Reactor Trip.

o As the Intermediate Range NIS Channel N-35 tracked NIS Channel N-36 for its normal operating range and returned to normal approximately ten (10) hours after the trip, no immediate action was deemed necessary. This abnormality has been observed and researched extensively in the past in cooperation with the NSS vendor, Westinghouse. No technical basis has been identified as to why the 1E-11 idle current does not maintain indication at 1E-11 amps. Rochester Gas and Electric Corporation (RG&E) and Westinghouse concurred that the channel was operable and capable of performing all intended functions. Further evaluations of the response characteristics of NIS Channel N-35 will be performed during the 1992 Annual Refueling and Maintenance Outage.

o The "A" Feedwater pump seal water D/P switch tubing was unplugged and flushed completely. This included both the high side and low side tubing. In addition, similar tubing for the "B" Feedwater pump was also flushed completely.

o The "A" Feedwater pump seal water D/P switch high side and low side tubing connections were checked for evidence of seepage and serviceability, and one section of tubing and connections was replaced.

o EWR 4960 was designed and scheduled for installation during the upcoming 1992 outage. The installation schedule was revised, and this modification was installed as a result of this transient. The feedwater pump seal water D/P switch time delay relays for both pumps were replaced, and the time delay setting was changed from five (5) seconds to a new setting of sixty (60) seconds. This modification provides a better opportunity for the operators to prepare for and mitigate a transient resulting from an impending trip of a MFW pump. In addition, the increased time delay does not compromise reliable operation of the MFW pumps on loss of seal water D/P, and also eliminates inadvertent MFW pump trips due to pressure surges and other short-term transients affecting the seal water D/P switch.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

As the underlying cause of the event was determined to be the plugging of the instrument tubing to the D/P switch because of a no flow condition, the following actions are being planned:

o Flushing will be performed on selected secondary system instrument tubing during the 1992 outage.

o Based on the results of flushing during the 1992 outage, a frequency of periodic flushing of selected instrument tubing will be established. The areas determined, if any, will then be put on a maintenance schedule.

VI. ADDITIONAL INFORMATION

A. FAILED COMPONENTS:

None

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events with the same underlying cause at Ginna Station could be identified.

C. SPECIAL COMMENTS:

None.

ATTACHMENT 1 TO 9204080122 PAGE 1 OF 1

RG and E
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March 30, 1991

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: LER 92-003, Feedwater Transient, Due to Loss of "A" Main Feedwater Pump, Causes Lo Lo Steam Generator Level Reactor Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

In accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (iv), which requires a report of, "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS), the attached Event Report LER 92-003 is hereby submitted.

This event has in no way affected the public's health and safety.

Very truly yours,

Robert C. Mecredy

xc: U.S. Nuclear Regulatory Commission
Region I
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Ginna USNRC Senior Resident Inspector

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